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# CAREM PROJECT DEVELOPMENT ACTIVITIES

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## ABSTRACT

The CAREM project involves technological and engineering solutions, as well as several innovative design features that must be properly demonstrated during the design phase. Also specific codes used for modelling systems related with safety issues to obtain design parameters (e.g. primary cooling system, reactor core, fuel design, etc.) must be verified and validated against world-wide benchmark and/or experimental data to build confidence on their results. This paper describes main issues of the development program ongoing as part of the design phase of the CAREM project, which includes the design and construction of several experimental facilities and engineering mock ups. Main results obtained from the test facilities and validation of codes are also presented.

**Keywords:** Integrated advanced reactor, Generation IV, inherent safety, passive safety systems, natural circulation reactors, Small and medium sized reactors (SMR).


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## 1 INTRODUCTION

The Argentinean CAREM project [1], which is jointly developed by CNEA and INVAP, consists on the development, design and construction of an advanced, simple and small Nuclear Power Plant (NPP).

CNEA has an extensive experience in basic research and nuclear related technology in several areas. This includes nuclear fuel cycle (from uranium mining and U-enrichment to fuel manufacturing), waste management and disposal, production and uses of radioisotopes, food irradiation, technology of nuclear materials, nuclear I&C, operation and maintenance of research reactors, nuclear and radiological safety, etc.

INVAP has been involved as designer and constructor of a wide range of technological projects. This includes research reactors, radioisotopes facilities, uranium enrichment, zirconium and beryllium processing, nuclear medical equipment, industrial waste treatment and disposal, environmental engineering and satellite construction among others.

So, the idea of design cycles has been applied in different frameworks involving several steps from the conceptual design to the final product (system, equipment, design code or technology process) capable of meeting the specific requirements. From early stages of CAREM project, engineering is underway in a globally planned sequence part of this design cycle, where two general stages may be recognised:

(a) Conceptual / basic design, and experimental activities as an aid to design.
(b) Detail design, and experimental activities for validation / qualification.

Within CAREM Project, the effort has been focused mainly on the nuclear island (inside containment and safety systems) where several innovative design solutions require developments of the first stage (to assure they comply with functional requirements). This comprises mainly: the Reactor Core Cooling System (RCCS), the Reactor Core and Fuel Assembly, internals of the Reactor Pressure Vessel (RPV), and the First Shutdown System (FSS). An extensive experimental plan has been prepared, including the design and construction of several experimental facilities to fulfil the Project's requirements.

An effort is planned for the systems/devices that require developments limited to the 2nd stage of a Design Cycle (qualification, or need of adaptation of a proven solution). I.e. they are not actually innovative by their features, but requires development effort in order to fit in the Project Engineering.

The RCCS modelling and qualification are boosted by the tests performed in a High Pressure Natural Circulation Rig (CAPCN), covering Thermal Hydraulics (TH), reactor control and operating techniques. The CAPCN rig reproduces all the dynamics phenomena of the RCCS, except for 3-D effects.

The Core Design involves different aspects i.e. study of thermal limits, neutronic modelling, structural mechanical and fuel assembly design. Neutronic modelling needs may be covered by benchmark data available world-wide and by experimental data from the Critical Facility RA-8. As for Fuel Assembly Design, CNEA has vast experience in the technology of nuclear fuels and structural and hydrodynamic tests will be carried out at Low and High pressures rigs.

The mechanical design of the core (structural, dynamic, seismic, etc.) and other RPVI, mock-up facilities are being constructed. They represent sections of the core, and include one vertical full-scale model with supporting Barrel and its Kinematics Chain.

The FSS, or more specifically the Control Rod Drives (CRD), is a good example of an innovative device design, comprising all the design cycle stages. An experimental plan is underway for the design and qualification stages.

The following is a brief description of some of the most relevant development tasks and their facilities that are carried out or foreseen as part of the CAREM project:


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## 2 DYNAMYC TESTS OF RCCS

The purpose of the High Pressure Natural Circulation Rig: CAPCN (figure 1), is mainly to study the thermal-hydraulic dynamic response of CAREM primary loop, including all the coupled phenomena that may be described by one-dimensional models. This includes the validation of the calculation codes on models of the rig, and the extension of validated models to the analysis of the CAREM reactor. The main tool used in thermal-hydraulic calculations is RETRAN-02.

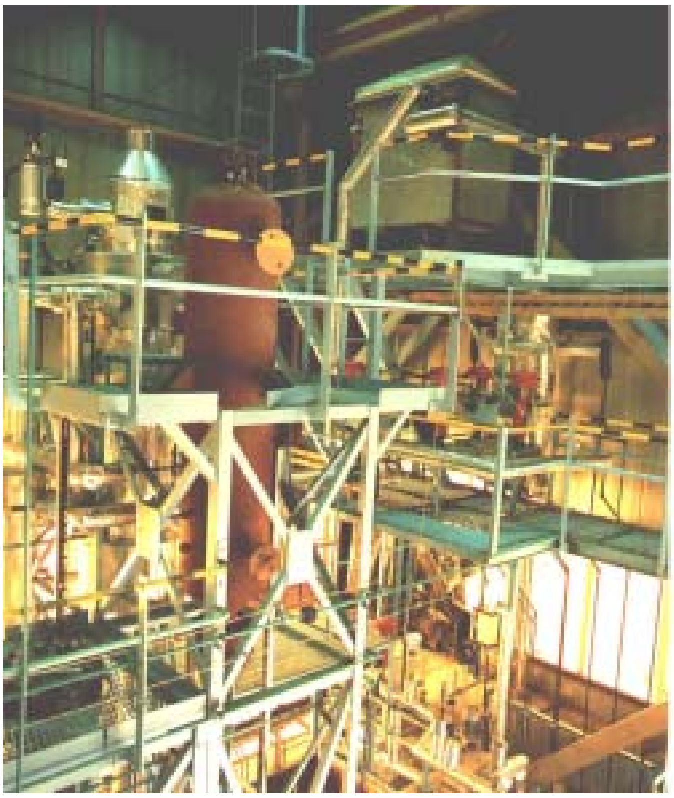

*Figure 1: CAPCN General View*

The CAPCN rig (see scheme figure 2) resembles CAREM in the primary loop (self-pressurised natural circulation) and the steam generator (helical once through), while the secondary loop is designed only to produce adequate boundary conditions. Operational parameters are reproduced for intensive magnitudes (pressure, temperature, void fraction, heat flux, etc.) and scaled for extensive magnitudes (flow, heating power, cross-sections, etc.). Height was kept approximately on a 1:1 scale.


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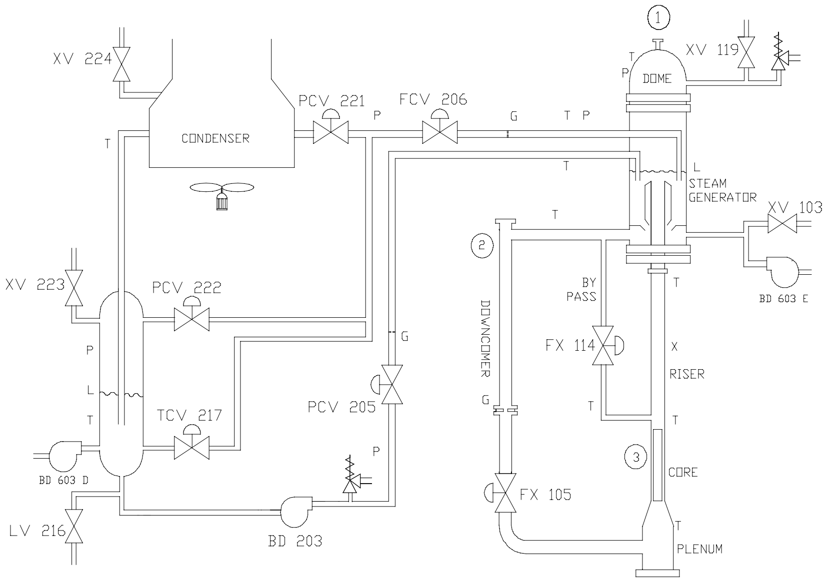


*Figure 2: CAPCN Simplified Process and Instrumentation Diagram*

The heating power may be regulated up to 300 kW, by the operator or by a feedback loop on primary pressure as a plain PID. An alternate feedback loop simulating core (neutronic) dynamics is under development.

The secondary loop pressure and cold leg temperatures are controlled through valves. The pump regulates the flow. The condenser is of air-cooled type with airflow control.

The control of the actuators (heaters, valves, pumps, etc.), data acquisition and operating follow up are carried out from a control room, through a PC based, multi-node software (flexible enough to define any feedback loop).

Most of the test [2] consist of an initial self-steady state in which a pulse-wise perturbation induces a transient. In this case the perturbation is a thermal unbalance as severe as possible for operational transients: thermal power is increased 12 KW (about 5% of FP) during 150 seconds. Primary pressure and circulating flow evolve mildly, with increases below 2 and 3% respectively, and primary temperatures hardly notice the perturbation. Therefore steam generation remains quite stable during the whole transient, a remarkable feature for a Steam Supply System (figure 3).

## 3 CHF TESTS AND THERMAL LIMITS

The TH design of CAREM reactor core was carried out using an improved version of 3-D, two fluid model THERMIT code. In order to take into account the strong coupling of the thermal-hydraulic and neutronic of the core, THERMIT was linked with the neutronic code CITVAP. This coupled model allows the "drawing" of a 3-D map of power and thermal-hydraulic parameters at any stage of the burn-up cycle.

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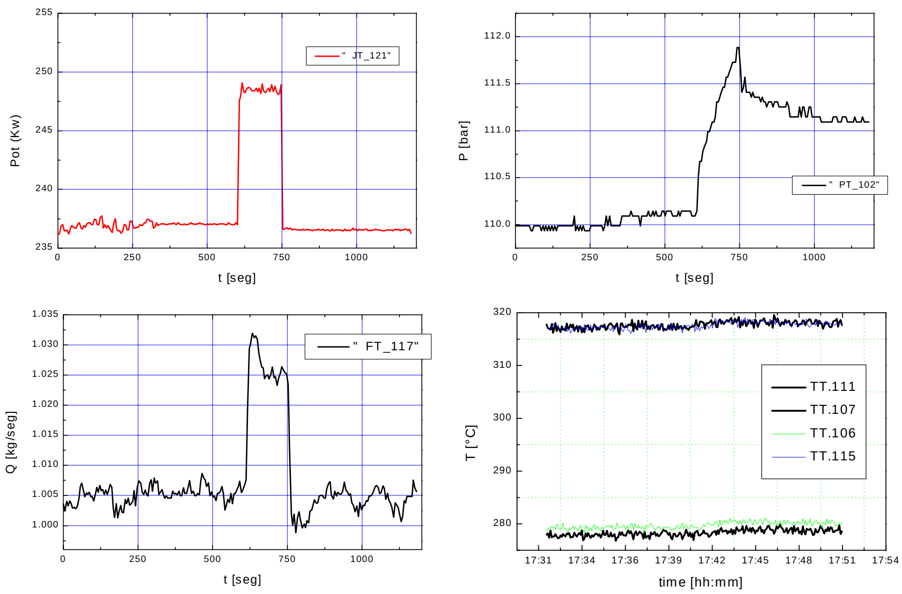


*Figure 3: Transient 1: +12 kW perturbation during 150 seconds*

The prediction of the thermal-limits (to harmful phenomenon like critical heat flux) of the fuel elements during operation and transients is considered of the utmost importance. Mass flow rate in the core of the CAREM reactor is rather low compared to typical light water reactors and therefore correlation or experimental data available are not completely reliable in the range of interest. Thus analytical data must be verified by ad-hoc experiments.

The experiments were conducted at the thermal-hydraulic laboratories of the Institute of Physics and Power Engineering (IPPE, Obninsk, Russian Federation). Figure 4 and 5 shows a view and scheme of the test section.

The main goal of the experimental program [3] was to generate a substantial database to develop a prediction methodology for CHF applicable to the CAREM core, covering a wide range of T-H parameters around the point of normal operation, i.e.:

| | | |
|---|---|---|
| ***Pressure*** | 10 – 13 | MPa |
| ***Mass Flux*** | 200 –700 | kg/m$^2$/seg |
| ***Quality*** | > -0.10 | |


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*Figure 4: Test Channel with the 19-rod bundle simulator*

Most tests were performed using a low-pressure Freon rig, and results extrapolated to water conditions through scaling models. Finally a reduced set of tests were performed in water at high pressure and temperature, to validate the method for scaling.

Different test sections were assembled to simulate different regions in the fuel element as well as radial uniform and non-uniform power generations. A bundle with 35 % of the full length was tested to obtain CHF data under average sub-cooled conditions. More than 250 experimental points under different conditions were obtained in the Freon loop and more than 25 point in the water loop.

The preliminary analysis of results from Freon loop measurements show that some existing correlations present quite a good agreement.

## 4 FUEL ASSEMBLIES

The developments tasks on this subject comprise mainly the two following issues:

- The improvement and extension of simulation models of BACO computational code, that may be considered under stage one of the Design Cycle.
- The verification, evaluation and qualification of the designs, as a development under stage two of the Design Cycle.

The BACO code [4] produces a best-estimate computer simulation of principal thermal-mechanical phenomena that occur within a nuclear fuel rod during burn-up process. It includes fission products generation and migration, fission gases release, in-clad pressure build-up, pellet deformation, crystallographic grain growth, stresses evaluation, pellet-clad interaction, etc.

This code had already been developed and verified against data of Fuel Assemblies of PHWR produced in Argentina. In order to cover Uranium Enriched Fuel Assemblies some new models had to be introduced and others had to be modified. These include the influence of high burn-up on thermal conductivity of UO2, the thermal conductivity in the pellet-clad gap (influence of Xe at high burn-up) and the migration of porosity (densification y restructuring).

These new models were validated through the participation in a Co-ordinated Research Project (CRP) of the International Atomic Energy Agency [5]. This CRP, called FUMEX Program, produces validation by initially sharing the experimental information of operating conditions and requirements of a certain fuel, and comparing "blind simulation" results with experimental measurements.

BACO code, combined with International Fuel Performance Experiments Database (of the OECD Nuclear Energy Agency) should cover the validation and evaluation requirements of the fuel rod design.


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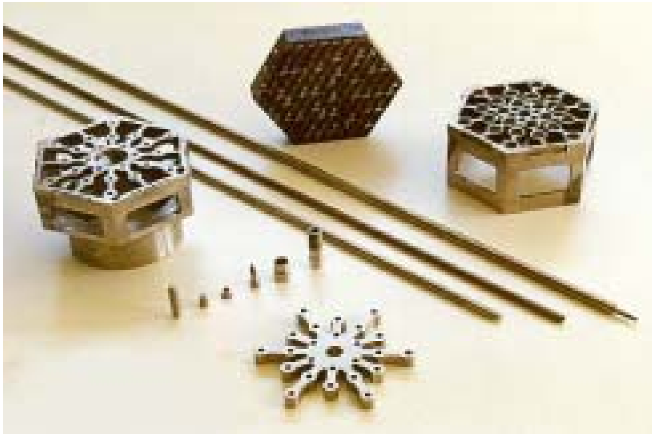

*Figure 5: Main fuel elements components during development stage*

The fuel assemblies and absorbing clusters (Figure 5) will be subject to a series of qualification tests, including standard mechanical evaluations, and hydraulic tests. The latter comprise:

- Tests in a Low Pressure Rig evaluating pressure-losses, flow-induced vibrations and general assembling behaviour.
- Endurance tests in a high pressure loop points to wear-out and fretting issues.

## 5 NEUTRONIC MODELLING VALIDATION

Validations against VVER reactors geometry [6, 7, 8] were made using experimental data from a ZR-6 Research Reactor, Central Research Institute for Physics, Academy of Sciences, Hungary, A series of benchmark data were used for typical PWR reactor. Further experimental data will be obtained from Critical Facility RA-8, which resembles certain CAREM neutronics issues. The neutronic calculation line used for the CAREM reactor core, the nuclear data library, and the validations that have already been made, are presented.

Nuclear Data: HELIOS library (190, 89 and 34 groups) used by the CONDOR code, has been especially developed to a group-wise (not an isotope-wise) order. Almost all the data in the library are based on the ENDF/B-VI data files.

Cell Code CONDOR 1.3: This code has the capability to calculate nuclear fuel elements with its spatial detail (without homogenisation). Collision Probabilities method (CPM) is used in a general 2-D cylindrical geometry. The Carlvik method with the macroband algorithm is applied to obtain the CP by the double numerical integration. The program uses normalisation schemes on integration chords (it preserves the regions volumes and surfaces) and CP (it preserves the reciprocity and balance between them).

The code CONDOR has been validated against VVER and PWR critical experiments in different conditions (fuel, moderator, $H_2O/U$ ratio, etc.). The validation against 26 VVER-type cells gave the result $K_{eff}=0.9953 \pm 0.0089$. Also the validation against 110 cases of $UO_2$ systems gave $K_{eff} = 1.00065 \pm 0.0080$ and over 86 cases of U-metal systems gave $K_{eff}=1.0044 \pm 0.0043$.

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Core Code CITVAP: This is a code derived from the well known diffusion code CITATION II, adding the following options using macroscopic cross-sections: Burnup, Fuel Management and Positioning of control rods. Preserving the entire original CITATION options, the most relevant features implemented in CITVAP are the following:

- Operation follow-up capabilities (fuel management and rod movements).
- Greater versatility in the input data improving 3 D problems description and makes it easier to describe fuel management and fuel element movements.

The code has been validated against VVER cell type reactors giving Keff=0.997±0.003 and against MTR reactors giving Keff=1.004±0.006.

## 6 HYDRAULIC CRD TESTS

One of the most innovative systems behind the CAREM concept is the Hydraulic (in-vessel) Control Rod Drive mechanism HCRDM. Two designs are under development: “Fast Extinction” and “Adjust & Control” CRD, being the latter that presents major challenges related to the design (Figure 6.).

The design embraces mechanical and thermal-hydraulic innovative solutions so feasibility of the concept must be demonstrated as a first step to be included in the reactor engineering.

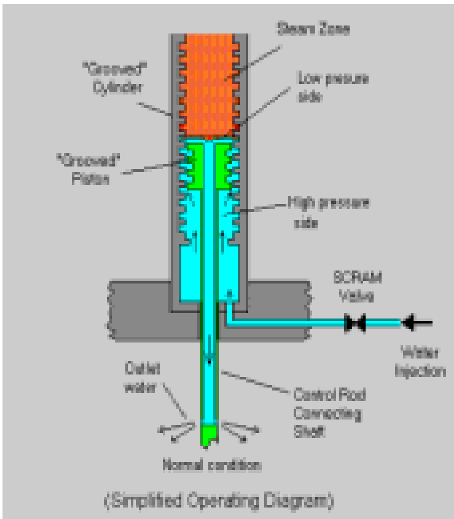


*Figure 6: HCRDM Adjust and Control System*

In other hand, their operational functions (the Adjust and Control, and the Fast Extinction) are part of one of the most important safety systems of the reactor: the First Shutdown System (FSS). These two features mean that a complete experimental program included “experiment-aided design” and qualification tests, is necessary to achieve the high reliability performance jointly with low maintenance.

The development plan refers to four well separate stages and includes the construction of several experimental facilities, reaching the testing of the system performance under RPV operating conditions. The four different stages and their (built or foreseen) facilities are:

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### 6.1 Preliminary tests (conceptual verification)

The aim of this test was to prove the feasibility of the theoretical approach, to have a first idea of some of the most sensitive controlling parameters and to determine spot points to be focused during design. Tests were undertaken on a rough device with promising experimental results, and good agreement with first modelling data was obtained.

### 6.2 First prototype tests

This stage pointed to determining preliminary operating parameters on a full-scale mechanism as a first approach towards detail engineering. These parameters include range of flow, ways to produce hydraulic pulses, etc. Manufacturing hints that simplified and reduce costs of the first design were also found. Tests were carried out in a craftily built rig and as part of this experimental development it was decided to separate the regulating and fast-drop requirements in different devices.

### 6.3 Test on a low pressure loop

This stage was carried out with the CRD at atmospheric pressure, and with feed-water temperature regulation up to low sub-cooling. The feed-water pipeline simulated alternative configurations of the piping layout with a second injection line (dummy) to test possible interference of pulses.

The ad-hoc test loop (CEM, Circuito de Ensayo de Mecanismos, figure 7) was designed to allow automatic control of flow, pressure and temperature, and its instrumentation produces information of operating parameters including pulse shape and timing. The tests included the characterisation of the mechanism and the driving water circuit at different operating conditions, and the study of abnormal situations as increase in drag forces, pump failure, loss of control on water flow or temperature, saturated water injection, suspended particle influence, and pressure “noise” in feeding line.

*Figure 7: Low Pressure Mechanism Test Rig*

The tests carried out at turbulent regime, which are the closest conditions to operation obtained in this loop, showed good reliability and repetitiveness as well as sensitivity margins for the relevant variables within control capabilities of a standard system.

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## 6.4 Qualification Tests

A high-pressure loop (CAPEM, Circuito de Alta Presión para Ensayo de Mecanismos) is being designed in order to reach the actual operating conditions (P= 12.25Mpa, T≈326°C). The main objectives are to verify the behaviour of the mechanisms, to tune up the final controlling parameter values and to perform endurance tests. After this stage, the system under abnormal conditions, such as the behaviour during RPV depressurisation, simulated breakage of feeding pipes, etc. will be tested.

# 7 RPV INTERNALS TESTS.

The mechanical structure of the core, supporting guides and all parts of the cinematic chain of the First Shutdown System are of particular interest. Complex assemblies and structures like the Steam Generator Units or ad-hoc mechanical solutions require the evaluation of manufacturing and assembly process, before finishing the design stage.

In sum, internals must be verified in order to define manufacturing, assembling allowances, and other detail engineering parameters to comply with their function during the RPV lifetime. Most tests are performed on mock-up facilities at 1:1 vertical scale. The following is a brief description of some of these devices, experimental plans and current status.

## 7.1 Full Scale Core-Sector

This is a complete-vertical representation of the core up to an extension of three fuel elements (i.e. structure, upper and lower grids, dummy FE, absorbing element guides, etc.) and major devices involved (i.e. absorbing fuel rods and connecting bar). All the structure can be perturbed by a hydraulic-driven actuator, which simulates minor vibrations and horizontal seismic loads on a wide range of frequencies and magnitudes (figure 8).

The aim is to make fine adjustments to the design, to verify couplings and auxiliary tools and to give a glance at the vibration modes of the whole assembly.

*Figure 8: Full Scale Core-Sector*

First series of experiments, have already been completed with encouraging results: the insertion of absorber rods, both in stepwise movement and rapid fall, was not affected by the perturbation of the whole dummy in a broad range of frequencies.

Also improvements in the guide devices for absorbing elements showed better performance while reducing manufacturing complexity from previous design. A second series of experiments to be performed under water are already planned.


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### 7.2 Full Scale (Vertical) Structural Barrel, Core and Cinematic Chain:

An important series of experiments to verify structural and dynamic behaviour of the Barrel will be performed after finishing those at the core sector. Being this structure very slender, the experiments deal with alignment, clearances in linear bearings, mass momentum and dynamic analysis. The latter point to determine natural frequencies, mode shapes and responses of the system under various external perturbations, which resemble seismic conditions and other vibrations.

The facility contains a complete sector representation (up to three fuel elements), similar to the Full Scale Core-Sector, including a sector of the Barrel and the Bar Guide Column (BGC) and one complete CRD System (hydraulic mechanism and all related parts of the cinematic chain including the absorbing element).

The whole device will be used to verify the performance of the Fist Shutdown System under normal and abnormal circumstances (i.e. misalignments or seismic) measuring the total elapsed time for rod dropping after SCRAM triggering. The vertical full scale will help to evaluate refuelling and maintenance manoeuvres, which have to be done underwater and far from the core region.

## 8 IN VESSEL INSTRUMENTATION

Since the HCRD design adopted has no movable parts outside the RPV, it was necessary to design a special probe to measure the rod position able to withstand primary environmental conditions. The proposed design consists in a coil wired around the HCRD cylinder with an external associated circuit that measure electric reluctance variations induced by the movement of the piston-shaft (made of magnetic steel) inside the cylinder.

Cold test performed showed that the system is capable of sensing one step movement of the regulating CRD, with an acceptable accuracy. In-furnace high temperature tests are going to be conducted to evaluate the behavior of the system against temperature changes similar to those occurring during operational transients.

The design of special high pressure removable feedthrouhs to allow dozens of electrical signals passing through RPV cover is also under work. This means the development and qualification of specific manufacturing and welding techniques.

## 9 CONCLUSIONS

The development program of the CAREM project has been driven to demonstrate the robustness of the design as well as being an irreplaceable tool to help the designers in issues related with the effectiveness and reliability of systems and components important to safety.

Relevant progress has been obtained for the validation of thermalhydraulic, neutronics and fuel codes using benchmark and experimental data.

Encouraging experimental results has been obtained towards the validation of the in-vessel hydraulic CRD design and the construction of a high pressure facility to perform tests at operating conditions is foreseeing.

Future work on development activities will be mainly focused on qualification of safety systems as well as manufacturing process for non-commercial or non-well qualified components, as those required by licensing program.


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